Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

MAY 11 2004

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Serial No.	04-282
MPS Lic/RWM	R0
Docket No.	50-336
License No.	DPR-65

DOMINION NUCLEAR CONNECTICUT, INC.
MILLSTONE POWER STATION, UNIT 2
LICENSEE EVENT REPORT 2004-002-00
AUTOMATIC REACTOR TRIP ON LOW STEAM GENERATOR LEVEL RESULTED FROM A FEEDWATER PUMP TRIP DURING TEST

This letter forwards Licensee Event Report (LER) 2004-002-00, documenting an event that occurred at Millstone Power Station, Unit 2, on March 15, 2004. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A), as an event or condition that resulted in an actuation of a reactor protection system.

If you have any questions or require additional information, please contact Mr. David W. Dodson at (860) 447-1791, extension 2346.

Very truly yours,

J. Alan Price
Site Vice President - Millstone

IE22

Attachments: (1)

Commitments made in this letter: None.

cc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. V. Nerses
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Mr. S. M. Schneider
NRC Senior Resident Inspector
Millstone Power Station

Attachment 1

Millstone Power Station, Unit No. 2

LER 2004-002-00

Dominion Nuclear Connecticut, Inc.

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
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TITLE (4)
Automatic Reactor Trip on Low Steam Generator Level Resulted From a Feedwater Pump Trip During Test

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	15	2004	2004	002	00	05	11	2004	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	x	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David W. Dodson, Supervisor Nuclear Station Licensing	860-447-1791

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

SUPPLEMENTAL REPORT EXPECTED (14)		EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

At approximately 2020 hours on March 15, 2004, with the unit in Mode 1 at 100 percent power, the reactor experienced an automatic reactor trip on low steam generator (SG) water level. The decreasing SG water level was due to a trip of the 'B' steam generator main feedwater (SGFW) pump that had occurred at approximately 2018 hours. The SGFW pump tripped in the course of performing the quarterly overspeed lockout test.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A) as any event that resulted in manual or automatic actuation of any of the systems listed in 50.73(a)(2)(iv)(B). This includes Reactor Protection System actuation (RPS) and Auxiliary Feedwater System initiation (AFW).

It is believed that the use of the lockout control switch for the quarterly performance of a SGFW pump emergency governor and trip lockout exerciser test is problematic and resulted in a momentary opening of switch contacts. This caused a rapid depressurization of the hydraulic overspeed protection system and a trip of the 'B' SGFW pump.

To prevent recurrence, the performance of applicable sections of the emergency governor and trip lockout exerciser test procedure at power levels that could challenge a reactor trip from a SGFW pump trip has been suspended until such time as an appropriate final resolution to this condition is specified.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

1. Event Description

At approximately 2020 hours on March 15, 2004, with the unit in Mode 1 at 100 percent power, the reactor experienced an automatic reactor trip on low steam generator (SG) water level. The decreasing SG water level was due to a trip of the 'B' steam generator main feedwater (SGFW) [SJ] pump that had occurred at approximately 2018 hours. The SGFW pump tripped in the course of performing its quarterly emergency governor and trip lockout exerciser test. Operators reset the tripped main feedwater pump but SG levels did not recover in time. The lowest observed SG level was 15 percent as compared to the normal level of 65 percent.

All control rods inserted into the core and electrical busses transferred properly following the reactor trip. Emergency diesel generators were operable, but not required. Auxiliary Feedwater (AFW) [BA] initiated as expected following the loss of a feedwater pump. Decay heat removal was established using the AFW system to feed the SGs, the atmospheric dump valves, and by bleeding steam to the main condenser through the condenser dump valves.

Following the reactor trip, the standard post trip procedure actions were carried out. The plant was monitored and the crew transitioned to the post trip recovery procedure. Safety functions status checks were performed successfully and, with the plant stable in MODE 3 (HOT STANDBY) at approximately 2200 hours, the crew transitioned to the normal operating procedure set.

This event is being reported pursuant to 10 CFR 50.73(a)(2)(iv)(A) as any event that resulted in manual or automatic actuation of any of the systems listed in 50.73(a)(2)(iv)(B). This includes Reactor Protection System actuation (RPS) and Auxiliary Feedwater System initiation (AFW). A non-emergency notification was made on March 15, 2004, to the NRC Operations Center (Event Number 40591) in accordance with 10 CFR 50.72(b)(3)(iv)(A) and 50.72(b)(2)(iv)(B).

2. Cause

The direct cause of lowering SG water levels and the subsequent reactor trip was the unexpected trip of the 'B' SGFW pump. The pump tripped during the quarterly testing of its emergency governor system. The controls required for the testing are located on the front standard of the SGFW pumps. When these controls are operated, they allow the testing of the hydraulic overspeed trip mechanism without actually tripping a running SGFW pump turbine. If the lockout control switch is released during testing, an actual trip of the SGFW pump will occur.

It is believed that the use of the lockout control switch for the quarterly performance of a SGFW pump emergency governor and trip lockout exerciser test is problematic and resulted in a momentary opening of switch contacts. This caused a rapid depressurization of the hydraulic overspeed protection system and a trip of the 'B' SGFW pump. The lockout control handswitch is a General Electric, SB-1, two position control switch that spring returns from the right position to center. The switch is configured for 90-degree travel between the two positions. When held in the LOCKOUT position, however, testing showed the sensitivity of the handswitch, and that it requires only a slight movement of approximately 5 to 7 degrees of travel to cause the switch to come out of the LOCKOUT position. In addition, and with the switch remaining in the LOCKOUT position, a twisting of the switch handle was also shown to cause contacts to open.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

3. Assessment of Safety Consequences

The purpose of the SGFW system is to provide water to the SGs for the transfer of thermal energy from the primary side of the SGs to the secondary side. The SGFW system is not safety related. The risk associated with this reactor trip is generally considered the same as from any normal plant trip since the Main and Auxiliary pumps remained available for decay heat removal. During and after the plant trip, there were no challenges to design basis event mitigation functions and no loss of any credited safety function from structures, systems and components. Consequently, this event is considered to be of low safety significance.

4. Corrective Action

Corrective actions are being addressed in accordance with the Millstone Corrective Action Program. Prior to use of the 'B' SGFW pump in support of unit restart, action was taken to repeat the testing to evaluate the reliability of installed components. To prevent recurrence, the performance of applicable sections of the emergency governor and trip lockout exerciser test procedure at power levels that could challenge a reactor trip from a SGFW pump trip has been suspended until such time as an appropriate final resolution to this condition is specified.

5. Previous Occurrences

No previous similar events/conditions were identified.

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].